ACCESSION #: 9612130137

LICENSEE EVENT REPORT (LER)

FACILITY NAME: Nine Mile Point Unit 1 PAGE: 1 OF 7

DOCKET NUMBER: 05000220

TITLE: Reactor Scram Caused by the Main Generator Lockout Relay

Trip

EVENT DATE: 11/05/96 LER #: 96-011-00 REPORT DATE: 12/05/96

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:

50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:

NAME: Mr. Kenneth Sweet, Manager TELEPHONE: (315) 349-2462

Technical Support, NMP1

COMPONENT FAILURE DESCRIPTION:

CAUSE: X SYSTEM: TL COMPONENT: CBL5 MANUFACTURER: G084

REPORTABLE NPRDS: N

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On November 5, 1996, at 2253 hours, Nine Mile Point Unit 1 (NMP1) received an automatic scram initiation signal resulting in a full reactor scram. The immediate cause of the scram was a trip of the main Generator Lockout Relay (86G1 Relay) which tripped the turbine and resulted in a reactor scram. At the time of the event, the plant was operating at 100 percent of rated thermal power. No testing or plant evolutions were in progress that resulted in or contributed to the scram. Subsequent to the scram, feedwater flow control valve #12 indicated closed but leaked excessively, causing water intrusion

into the main steam lines and emergency condenser lines. The cause of the Generator Lockout Relay (86G1 Relay) activation is a failed flexible link in the generator exciter. The link severed and contacted ground initiating a ground fault. A failure analysis will be performed on the failed component.

The immediate corrective actions were to perform scram recovery actions, place the plant in a stable condition, and determine the cause of the scram. Subsequently, an evaluation was performed of the impact of water in the main steam lines, the failed flexible link was replaced, the feedwater flow control valve 12 was rebuilt and tested to ensure minimal leakage, and an evaluation of the emergency condenser piping movement was performed.

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I. DESCRIPTION OF EVENT

On November 5, 1996, at 2253 hours, Nine Mile Point Unit 1 (NMP1) received an automatic scram initiation signal resulting in a full reactor scram. The immediate cause of the scram was a trip of the Main Generator Lockout Relay (86G1 Relay) which tripped the turbine and resulted in a reactor scram. The 86G1 Relay trip was initiated by tripping of the 64F field ground relay which was due to a failed flexible link in the generator exciter. At the time of the event, the plant was operating at 100 percent of rated thermal power. No testing or plant evolutions were in progress that resulted in or contributed to the scram.

Prior to the event, the Feedwater System was in a normal operating lineup consisting of Condensate Pumps 11, 12, and 13, Feedwater Booster Pumps 11 and 13, and Feedwater Pumps 11 and 13. Feedwater Pump 12 was in standby mode. When the turbine tripped, water level lowered to +29 inches (9 feet 5 inches above the top of fuel) and the feedwater system transferred to the High Pressure Coolant Injection (HPCI) Mode of operation, and

Feedwater Pump 13 continued to provide flow as the main turbine coasted down. Feedwater Pump 12 started as designed.

With the three feedwater pumps running, vessel level recovered to normal level in approximately 30 seconds. As water level was returning to the normal band, low suction pressure caused Feedwater Pump 11 to trip. The cause of the low suction pressure trip of the feedwater pump has been attributed to the high rate of flow with all three feedwater pumps running and only five of six condensate demineralizers in service (one of six was out of service for planned maintenance). Water level continued to rise and overshot the normal HPCI control setpoint of 72 inches. At 95 inches, Feedwater Pumps 11 and 12 are designed to trip (unless their flow control valves are indicated fully shut) and Feedwater Pump 13 declutches. Feedwater Pump 13 declutched as designed. As noted, Feedwater Pump 11 had tripped earlier on low suction and Feedwater Pump 12 did not trip since the flow control valve indicated shut. Reactor water level continued to rise past 100 inches, which is the calibrated range of the narrow range level instrument used by the operator in monitoring normal reactor water level.

At this time, the operator monitoring reactor water level referenced the wide range water level instrument which indicates in units of feet. The Station Shift Supervisor made a decision not to trip the running Feedwater Pump 12 because the crew had not yet been able to determine the reason for the trip of Feedwater Pump 11 and did not want to secure the only available HPCI pump. The operator monitoring water level knew that water was approaching the main steam line nozzles and had established a meter reading of 11 feet at which immediate action would have been initiated to prevent water from entering the main steam lines. The elevation of the lower edge of the main steam lines is 11.7 feet. Water level on the wide range meter leveled off at less than 11 feet indicated. The operator monitoring level took no additional action since, based upon indicated level, water was below the main steam line nozzles. However, due to an inaccuracy (apparently one foot low) in the wide range instrumentation while at normal operating temperature and pressure (this instrument is calibrated for cold operating conditions), water entered the main steam lines

I. DESCRIPTION OF EVENT (cont'd)

causing damage to one hydraulic snubber (03-HS-02) and one rod hanger in the steam bypass line to the main condenser. After approximately 42 minutes, water level began a decreasing trend due to reactor water cleanup reject flow.

Evidence of emergency condenser pipe movement was identified during post-scram plant walkdowns. This is attributed to the duration and magnitude of the reactor vessel high water level condition and subsequent draining back to the reactor pressure vessel. However, engineering concluded that based upon their walkdown that no damage occurred to the piping or piping supports

The turbine trip from high power caused a pressure transient which resulted in four of the six Electromatic Relief Valves (ERVs) opening as expected, which caused torus water temperature to rise 1.1 degrees Fahrenheit. The longest duration of an ERV lift was four seconds.

The reactor scram was reset at 2259 hours.

II. CAUSE OF EVENT

A root cause evaluation of the generator trip was commenced per NMPC procedure NIP-ECA-01, Deviation/Event Report. The immediate cause was a failed flexible link in the generator exciter. The flexible link severed and contacted ground initiating the ground fault. This flexible link was disconnected for routine exciter maintenance in 1993 and reinstalled. At that time, no abnormalities were noted. A failure analysis will be performed on the failed component.

The cause of water level rising above the HPCI trip point of 95 inches and entering the main steam line is excessive leakage past feedwater Flow Control Valve 12. The flow control valve close limit switch was actuated, and indicated valve closure and prevented a pump trip. However, the valve leaked and continued to provide flow to the reactor.

The cause of the excessive leakage past feedwater Flow Control Valve 12 was attributed to valve setup based upon new diagnostic testing which was conducted. The testing revealed that with a full closure signal the valve would not be firmly seated closed.

The reactor pressure vessel level remained above the bottom of the main steam line nozzles for approximately 42 minutes due to the following contributors: 1) In 1992, when the bias in the wide range instrumentation was identified there was inadequate compensatory action taken such as an operator aide. Consequently, operators lacked knowledge as to the limitation of the device. 2) The operating procedures, simulator

III. CAUSE OF EVENT

response, and training conditioned the operators to maintain HPCI pumps running even when above their prescribed level control band. Their response was conditioned more to establishing reject flow through reactor water cleanup and long cycle recirculation to the condenser rather than terminating HPCI injection.

III. ANALYSIS OF EVENT

This event is reportable in accordance with 10CFR50.73(a)(2)(iv), which requires Licensees to report "any event or condition that resulted in a manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS)."

The full reactor scram was the approved design response of the Reactor Protection System (RPS) to a main turbine/generator stop valve closure with the plant at 100 percent power. The ERVs lifted and reseated, as designed, to prevent an overpressure condition. The HPCI mode of the feedwater system initiated to maintain reactor vessel water level as designed. The feedwater pump high level control system functioned as designed. However, since the "closed" limit switch for flow control valve 12 was actuated, Feedwater Pump 12 did not receive a trip signal. With Feedwater Pump 12 running and the flow control valve leaking, water continued to flow into the reactor vessel and into the main steam lines. This reactor scram event was less severe than that which is bounded by the Electrical Load (Generator Trip) Transient Analysis in Chapter XV of the NMP1 FSAR.

A team of Niagara Mohawk engineers from mechanical design, structural design, and a system engineer performed a walkdown of the main steam lines and the main steam bypass lines to assure that there was no damage from the water intrusion which would render the system inoperable. The team's initial walkdown revealed that a snubber and a rod support on the main steam bypass line to the condenser were damaged as a result of the intrusion. The team also had non-destructive examinations performed on two high stress welds to verify no physical damage. The reactor main steam nozzles were evaluated and it was determined that the transient did not have an adverse impact based upon the heat capability of the vessel, the short duration and the small temperature difference. During startup, the team walked the systems down at 25 percent power to verify that the piping and piping support system functioned as designed. Their overall conclusion was that the systems are in full compliance with the design basis.

The pipe movement due to the water intrusion into the Emergency Condenser system was also evaluated by nuclear engineering personnel. A team of engineers walked the Emergency Condenser system down to verify that piping movement did not result in damage to the system, which would make the system inoperable. In addition, the team evaluated the cause of the piping movement. Based upon their expert opinion, the team concluded that the piping movement was caused by the formation and collapse of vapor pockets in the piping

III. ANALYSIS OF EVENT (cont'd)

system. Although there was minor damage to some insulation, there was no physical damage which would render the system inoperable. Engineering concluded that the system was in full compliance with the design basis.

There were no adverse safety consequences as a result of this event. No systems or components were inoperable that contributed to the severity of this event. The reactor scram posed no threat to the health and safety of the general public or plant personnel.

IV. CORRECTIVE ACTIONS

The immediate corrective actions were to perform scram recovery actions, place the plant in a stable condition, and determine the cause of the scram.

Additional corrective actions include:

1. Engineering completed an evaluation of the Main Steam and Emergency

Condenser systems which included inside and outside the drywell walkdowns. The conclusion is that the systems are in full compliance with their design basis.

2. In addition, non-destructive examinations were performed on high stress points of the Main Steam piping to ensure system integrity. The results were determined to be satisfactory.

3. The snubber and rod hanger on the steam bypass line to the condenser were repaired.

4. The flexible link was replaced and the failed link will undergo a failure mode evaluation by January 15, 1997. Depending upon the results, additional appropriate actions will be taken to minimize the potential for similar failures. Appropriate periodic preventive maintenance will be established for this conductor by February 15, 1997.

5. Feedwater Flow Control Valve 12 was rebuilt and tested to verify satisfactory performance. This rebuild also calibrated the valve close signal to ensure full closure when the valve controller calls for it.

6. Feedwater Flow Control Valve 11 was tested and a leak check performed to verify satisfactory performance. This included calibration of the valve close signal to ensure full closure when the valve controller calls for it.

IV. CORRECTIVE ACTIONS (cont'd)

7. The Instrument and Control procedure was revised to ensure that firm seating of feedwater flow control valves 11 and 12 would occur by reducing the low limit of the HPCI controllers.

8. An evaluation of feedwater level control following reactor scrams will be performed to determine enhancements to minimize the potential for water entering the main steam lines by June 30, 1997.

9. Based upon DER 1-96-3030 and associated operability determination performed on the wide range instrumentation, an operator aid was posted on the panel adjacent to the wide range water level meter to alert the operator of the indicated level discrepancy when the instrument is hot.

10. By January 15, 1997, a causal analysis will be completed to determine why adequate compensatory actions were not taken in 1992. Additional corrective actions will be taken as appropriate.

11. Operators were provided additional training on the wide range meter discrepancy and a procedure change has been made to enhance the operators' ability to take mitigating actions should water level approach the main steam lines. Supplemental training for the operators including simulator exercises for the condition will be developed and provided by December 31, 1996. After the training is completed, Operations and Training management will evaluate whether additional procedure and/or training changes are necessary by

January 31, 1997.

12. NMPC will evaluate the calibration methods and the source of the inaccuracies in the wide range level instrumentation to determine if the instrumentation can be performed to more accurately indicate reactor water level by February 15, 1997.

V. ADDITIONAL INFORMATION

A. Failed components: Flexible Link in Generator Exciter

B. Previous similar events: LER 95-02, "Reactor Scram Caused by Failure of Generator Protective Relay." On April 19, 1995, NMP1 received an automatic scram initiation signal resulting in a full reactor scram. The immediate cause of the scram was a trip of the Main Generator Lockout Relay (86G2 Relay) which tripped the turbine and resulted in a reactor scram. The 86G2 relay trip was initiated by failure of an oil-filled capacitor in a generator protective relay. The corrective actions taken for that event would not have prevented this event.

C. Identification of components referred to in this LER:

COMPONENT IEEE 803 FUNCTION IEEE 805 SYSTEM ID

RPS N/A JC

Turbine Generator TRB TA

Main Feedwater System N/A SJ

High Pressure Coolant Injection N/A BJ

Reactor Pressure Vessel N/A SB

Protection Relay 21 FK

Pumps P SJ

Relief Valves RV SB

Emergency Cooling System COND BL

Generator Lockout Relay 86 TL

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NIAGARA MOHAWK

G E N E R A T I O N NINE MILE POINT NUCLEAR STATION/LAKE ROAD,

BUSINESS GROUP P.O. BOX 63, LYCOMING, NEW YORK 13093

December 5, 1996

NMP1L 1166

U. S. Nuclear Regulatory Commission

Attn: Document Control Desk

Washington, DC 20555

RE: LER 96-11

Docket No. 50-220

Gentlemen:

In accordance with 10 CFR 50.73 (a)(2)(iv), we are submitting LER 96-11, "Reactor Scram Caused by the Main Generator Lockout Relay Trip."

Very truly yours,

Norman L. Rademacher

Plant Manager - NMP1

NLR/GJG/kap

Enclosure

xc: Mr. H. J. Miller, Regional Administrator

Mr. B. S. Norris, Senior Resident Inspector

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